NON-PUBLIC?: N
ACCESSION #: 9110170115
LICENSEE EVENT REPORT (LER)

FACILITY NAME: NORTH ANNA POWER STATION UNIT 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000339

TITLE: REACTOR TRIP CAUSED BY MFRV CLOSURE UPON FAILURE OF DRIVER CARD
AND SUBSEQUENT SAFETY INJECTION CAUSED BY MALFUNCTION OF THE STEAM DUMP CONTROL SYSTEM
EVENT DATE: 09/20/91 LER #: 91-009-00 REPORT DATE: 10/10/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: G. E. Kane, Station Manager TELEPHONE: (703) 894-2101

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: FCV MANUFACTURER: C635
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On September 20, 1991, at 0513 hours, with Unit 2 at 100 percent power, the "B" Main Feed Regulating Valve (MFRV) closed due to a driver card failure causing a reactor trip. The initiating signal for the reactor trip was "B" Steam Generator low level coincident with a steam flow greater than feed flow mismatch. After the reactor trip, an automatic Safety Injection (SI) was initiated. The SI was initially determined to be non-valid; however, a detailed investigation revealed that a malfunction in the steam dump control system resulting in a high steam flow with low-low Reactor Coolant System (RCS) average temperature generated the SI signal. This was the sixth SI to occur on Unit 2. This event is reportable pursuant to 10CFR50.73 (a) (2) (iv), and a four hour report was made pursuant to 10CFR50.72 (b) (2) (ii). A Notification of Unusual Event was declared upon determination that the SI signal was

valid.

The "B" MFRV closure and subsequent reactor trip was caused by failure of the valve driver card. The SI signal was caused by a malfunction in the steam dump control system which prevented the valves from modulating closed.

No significant safety consequences resulted from this event because all safety systems responded appropriately. Therefore, the health and safety of the public was not affected at any time during this event.

END OF ABSTRACT

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1.0 Description of the Event

On September 20, 1991, at 0513 hours, with unit 2 operating at 100 percent power (mode 1) the "B" Main Feedwater Regulating Valve (EIIS System Identifier SJ, Component Identifier FCV) closed due to a driver card failure causing a reactor trip. The initiating signal for the reactor trip was "B" Steam Generator low level coincident with a steam flow greater than feed flow mismatch. Subsequent to the reactor trip, an automatic Safety Injection (SI) was initiated. The SI was initially determined to be non-valid; however, a detailed post-trip data analysis revealed that a malfunction in the steam dump control system led to a high steamline flow with low-low average reactor coolant system (EIIS System Identifier AB) temperature condition which had, in fact, generated a valid SI signal. This was the sixth SI to occur on Unit 2. At 0520 hours the SI signal was reset in accordance with emergency procedures. This event is reportable pursuant to 10CFR50.73 (a) (2) (iv) as an automatic actuation of an Engineered Safety Feature. A four hour report was made at 0858 hours pursuant to 10CFR50.72 (b) (2) (ii).

A Notification of Unusual Event (NOUE) was declared and notifications made to state and local governments at 1050 hours following determination that although no accident condition existed, the SI signal was, in fact, valid. The NRC Operations Center was notified at 1054 hours, and the NOUE was terminated on the same call.

During the event, heat removal was initially accomplished by dumping steam to the main condensers through the steam dumps. The steam dump flow control valves are designed to modulate closed as RCS average temperature decreases to normal no-load values. These valves failed to modulate closed thus causing a high steam flow coincident with low-low RCS average temperature SI signal to be generated.

Control Room Operators responded to the event in accordance with Emergency Procedure E-0, "Reactor Trip or Safety Injection". RCS temperature and pressure decreased to 534 degrees F and 1829 psig before recovering to 553 degrees F and 2235 psig. Plant safety equipment responded appropriately during the reactor trip and SI.

After event investigation and corrective action, unit 2 was taken to critical on September 21, at 1020 hours.

2-0 Significant Safety Consequences and Implications

No significant safety consequences resulted from this event because all safety systems responded appropriately, and there was no significant release of radioactive materials. Therefore, the health and safety of the public was not affected at any time during this event.

3.0 Cause of the Event

The cause of the steam flow greater than feed flow reactor trip was a failed power supply transistor on the driver card for the "B" main feedwater regulating valve. When power was lost, the valve failed closed and isolated normal feedwater flow to the "B" steam generator.

The cause of the SI was a high steam flow condition coincident with low-low RCS temperature following the reactor trip. The failure of a card in the steam dump control system caused the system to continue to dump steam after no-load RCS average temperature was reached. This led to the rapid cooldown and subsequent SI.

4.0 Immediate Corrective Actions

Upon determination that "B" MFRV had closed, the operator attempted to open the valve using the controller on the benchboard. Since the card had failed, the controller would not respond. Control room personnel then responded to the reactor trip in accordance with Emergency Procedure E-0, "Reactor Trip or Safety Injection". The SI signal was reset at 0520 hours, and a Notification of Unusual Event was declared at 1050 hours when it was determined that a valid SI had occurred.

5.0 Additional Corrective Actions

The following corrective actions were taken to correct the hardware problems that occurred with the MFRV and steam dump control systems.

o The failed MFRV driver card was replaced with a new card and a successful functional test was performed.

o The failed card on the steam dump control system was replaced and satisfactorily tested.

6.0 Actions to Prevent Recurrence

An engineering evaluation and root cause investigation is being conducted to discern any contributing factors to the card failures and determine if other similar cards have an increased failure risk.

The emergency response procedures are being evaluated to determine if more specific directions are required concerning the declaration of a NOUE for an ECCS actuation.

The preventative maintenance program which tests the steam dump control logic will be evaluated to ensure proper measures are taken for failures which would cause the steam dumps to not respond properly during trips.

7.0 Similar Events

LER N1-90-001-00 documents a Unit 1 reactor trip from 100 percent power due to a failed driver card on the "C" MFRV. A root cause evaluation determined that some components on Westinghouse 7300 driver cards should be replaced every five years due to equipment aging. Upon determining that an aging problem existed, all driver cards for the MFRVS on Units 1 & 2 were replaced with refurbished driver cards during the last refueling outages.

LER N1-89-005-00 documents a Unit 1 reactor trip from 76 percent power due to fatigue failure of the Instrument Air supply line to the "C" MFRV.

8.0 Additional Information

Unit 1 was operating at 100 percent power throughout the event and was not affected.

Vepco VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402

MINERAL, VIRGINIA 23117 10 CFR 50.73

October 10, 1991

U. S. Nuclear Regulatory Commission Serial No. N-91-026
Attention: Document Control Desk NAPS: WCH
Washington, D.C. 20555 Docket No. 50-339
License No. NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 91-009-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours,

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. S. Lesser
NRC Senior Resident Inspector

North Anna Power Station

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